NON-PUBLIC?: N
ACCESSION #: 9108220114
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Brunswick Steam Electric Plant PAGE: 1
Unit 1

DOCKET NUMBER: 05000325

TITLE: Reactor Scram Resulting From Common Instrument Header Pressure
Perturbation
EVENT DATE: 07/18/91 LER #: 91-018-00 REPORT DATE: 08/16/91

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR
SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: Glen M. Thearling, Regulatory TELEPHONE: (919) 457-2038
Compliance Specialist

COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: BJ COMPONENT: PI MANUFACTURER: A501
X BLR ISV D232
X JM 33 N007
B JM ISV A391

REPORTABLE NPRDS: Y
Y
Y
Y

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On July 18, 1991, Unit 1 reactor was at 100% power when it scrammed as
the result of a Main Steam Line Isolation signal, inadvertently generated
during surveillance 1MST-RSDP-21Q for reactor water level transmitter
B21-LT-N026B, which was being returned to service. After extensive
investigation it is suspected that an instrument isolation valve
leaked-by, resulting in a pressure transient on the common instrument

header which has instruments feeding both reactor protection divisions. The resulting erroneous level signals caused the following system actuations: Isolations - Group 1 (Main Steamlines), Group 3B (1/2 Reactor Water Cleanup System), and Initiations - Reactor Core Isolation Cooling (RCIC), High Pressure Coolant Injection (HPCI), Standby Gas Treatment (SBGT), Core Spray (CS), Emergency Diesel Generators (EDG).

The Emergency Core Cooling Systems (ECCS) were operable.

The safety significance of this event is minimal as plant safety systems responded as required.

END OF ABSTRACT

INITIAL CONDITIONS

On July 18, 1991, Unit 1 was at 100% power and had been synchronized to the grid for 72 days. All Emergency Core Cooling Systems were operable. Surveillance 1-MST-RSDP-21Q was almost complete and the reactor vessel level transmitter B21-LT-N026B was being returned to service. The level transmitter B21-LT-N026B was isolated with the calibration equipment connected to its drain/calibration lines.

EVENT NARRATIVE

At 17:20 on July 18, 1991, while preparing to return the 1-B21-LT-N026B instrument transmitter to service a pressure transient on the common instrument variable leg header resulted in a reactor scram and Primary Containment Isolation System (PCIS) group isolations. The instrument variable leg drain valve was being opened to pressurize the instrument, when leakage by instrument manifold isolation valve 1-B21-LT-N026B-4 resulted in the pressure transient on the common instrument variable leg header, whose instruments feed both reactor protection divisions.

The 1-B21-LT-N026B transmitter (see attachment A) is connected to common variable and reference leg headers through root isolation valves (normally left open during calibrations), and instrument manifold block isolation valves (used to isolate the instrument during calibration). The instrument manifold block contains: both instrument transmitter isolation valves, an equalizing valve that can cross-tie the two headers at the transmitter, and a line from each side of the transmitter that through a pair of drain valves connect to the drain/calibration headers.

In accordance with the surveillance, the instrument reference leg drain valve was being opened to pressurize the instrument prior to opening the manifold isolation valves. This process minimizes the differential pressure across the isolation valve and should prevent a pressure transient on the common instrument header. The gauge connected to the drain/calibration header initially indicated 20 psig prior to the drain valve being opened. While opening the drain valve the technician became aware of the transient and had the second technician verify that his hand remained on the correct valve. It was also noted that the gauge on the drain/calibration header had increased to approximately 70 psig. This supports that the variable leg manifold isolation valve had leaked-by into the transmitter. When the transmitter drain valve was opened a path to depressurize the transmitter and perturbate the common instrument variable leg header through the leaking isolation valve was created. The pressure transient was sensed as an erroneous level decrease actuating PCIS and Emergency Core Coolant Systems (ECCS). The PCIS Group 1 isolation command closed all Main Steamline Isolation Valves (MSIV) and the PCIS Group 3B isolation command closed the Outboard Reactor Water Cleanup Isolation Valve. The following systems received start commands: CS, HPCI, RCIC, and the four EDG's. Due to the brevity of the erroneous level spike the RHR system did not have time to latch-in a start command. The MSIV closure generated a full reactor scram and all control rods fully inserted.

The combination of reactor power decrease and a closure of the MSIV's caused an actual momentary reduction in reactor vessel water level due to steam void collapse. PCIS Groups 2, 3, 6 and 10 isolations were received for the actual reactor vessel low level signals. The Group 8 isolation was present prior to the event due to normal operating reactor pressure and the associated valves were closed. Safety Relief Valves operated as designed to control reactor pressure. PCIS Isolations and ECCS initiations occurred as designed.

Additional problems identified during the event were:

1. A limit switch problem on the outboard MSIV 1-B21-F028B resulted in dual indication for two and one-half minutes after its closure.

2. An oil leak of approximately 0.435 gpm on the HPCI Turbine Oil Filter Inlet Pressure Gauge E41-PI-5549, drained about 10 of the approximately 88 gallons available during the 23 minutes prior to the leaks isolation. The leakage rate would have allowed another 45 minutes prior to the low level alarm and 60 minutes prior to a loss of oil pressure. If the oil leak had

not been isolated eventually the oil operated HPCI trip valve would have closed due to low oil pressure, and the HPCI turbine would have stopped prior to bearing damage. This failure is actually less significant than most HPCI failure modes in that, if HPCI had been needed, it would have allowed significant HPCI operation prior to the actual system loss.

3. The Main Steamline Drain Inboard Isolation Valve, 1-B21-F016 motor breaker tripped on thermal overload when it was being opened to equalize pressure around the MSIV's. This Anchor-Darling, 3-inch, double-disc gate valve when disassembled for inspection/repair was found to have cuts, at the same height, in the valve body's four disc guides caused by the valve discs. The height of the cuts, the direction of the deformed metal and the shape of the cuts showed that the valve discs had jammed while opening, at about 75% open. The root cause was determined to be the sharp edges on the valve discs. The sharp edges are a generic concern which is only applicable to the 3-inch Anchor-Darling double-disc gate valves. While this is a generic concern, it is not a generic Safety Issue because the safety function of each of the valves used on both Units is to close. The valves of concern are: Main Steamline Drain Isolation Valves 1/2-B21-F016 and 1/2-B21-F019, and RCIC Steamline Isolation Valves 1/2-E51-F007 and 1/2-E51-F008.

CAUSE OF EVENT

The pressure transient on the common variable leg instrument header, was caused by the leaking isolation valve (1-B21-LT-N026B-4) on Reactor Water Level Transmitter B21-LT-N026B. While the valve was found fully closed and testing could not recreate the transient it is felt that this is the only cause that the data collected during the investigation will support. Four possible causes for the event were exhaustively pursued:

1) For the 1-B21-LT-N026B-4, a valve failure mode was not found even though it was destructively tested.

2) If debris kept the 1-B21-LT-N026B-4 from properly seating closed it may have been flushed out during the transient and was therefore not recovered.

3) The possibility of the 1-B21-LT-N026B-4 not being fully closed during the initial valving out process was investigated, but due to the technicians use of double verification and that a wrench was used to insure full closure of the instrument isolation valves a human error was considered unlikely.

4) That human error resulted in a valve being repositioned out-of-sequence during the surveillance was considered unlikely, due to the technicians performing self checking and double verification prior to the valve operation. Additionally valves that could have been manipulated to cause this are not configured such that it is likely a wrong valve was repositioned.

CORRECTIVE ACTIONS

1) The isolation valve 1-B21-LT-N026B-4 was removed for testing and replaced along with the transmitter isolation valve manifold and both transmitter drain valves.

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2) The surveillance procedures 1/2MST-RSDP21Q will be revised to provide closure of root valves in order to provide a better isolation during future surveillances. (Due 8/30/91)

3) Prior to testing a review will be conducted of the other level calibration procedures performed at power, to identify those requiring root valve closure.

4) To minimize other possible valve leakage problems on this rack, an evaluation will be performed on the common instrument rack drain/calibration headers being be left vented. (Due 10/26/91)

5) To minimize other possible valve leakage problems on this rack, an evaluation will be performed for the common instrument rack drain/calibration headers permanent removal. (Due 11/27/91)

6) Review the Scram event with appropriate Instrumentation and Control (I&C) personnel prior to the next performance of the 1/2MST-RSDP21Q. (Due 8/22/91)

7) Review industry data related to the instrument valve leakage during similar events. (Due 9/26/91)

8) Unit 2's B21-LT-N026B valves will be inspected for leaks during the next scheduled outage. (Due 11/27/91)

9) During the next refueling outages the seven other 3-inch Anchor-Darling double-disc gate valves will be inspected. (Due 11/27/91 on Unit 2 and 12/6/92 on Unit 1)

10) Anchor-Darling has been requested to review the existing design for their 3-inch double-disc gate valves and provide suitable replacement discs with an improved edge configuration, if appropriate. (Due 9/11/91)

11) The Main Steamline Drain Inboard Isolation Valve 1-B21-F016 was temporarily repaired/tested with the raised metal on the four disc guides removed (to prevent possible foreign-object damage within the system), and the open limit switch reset to 60% so as to avoid the degraded guide area where jamming is possible. The safety function
which is to close,
is unaffected.

12) The MSIV 1-B21-F028B closed limit switch was adjusted.

13) A Technical Specification change will be evaluated to allow the remote shutdown panel reactor level instrument surveillances to be performed at a refueling frequency.

14) The HPCI oil pressure gauge has been isolated and placed under clearance pending replacement,

SAFETY ASSESSMENT

The instrument header pressure transient event posed minimal safety significance since all systems performed their safety related functions.

The HPCI oil leak was of minimal safety significance as the 60 minutes between the time the alarm would have come in and the system shutdown would have been adequate for Operations to prevent loss of the system by investigating the cause and closing the gauge isolation valve. If the HPCI room had been inaccessible, HPCI would have been lost after approximately 1 hour and 20 minutes. As HPCI is a single train system and is not single failure proof the availability of the Automatic Depressurization System (ADS) with the low pressure ECCS systems would have assured adequate safe shutdown capability for this failure mode. This failure is actually less significant than most HPCI failure modes in that it would allow significant HPCI operation prior to the actual loss of the system.

The failure of the Main Steamline Drain Inboard Isolation Valve 1-B21-F016 is not significant in that the safety function of the valve is to close (if open) during an accident. The valve has no specific safety function to open, and receives no automatic open signals. The inability to open this valve would prevent reestablishing the Condenser as a heat sink. While this is desirable, it is not required for a safe reactor shutdown. The other identical valves which are currently installed, also only have a safety function to close. Since the potential sharp edges of the discs for these valves would only affect the opening stroke, the safety function of these valves would not be compromised. The valve design problem is not reportable per 10 CFR 21, because of our plants application. Anchor-Darling has been notified of the problems we are having with this valve.

PREVIOUS SIMILAR EVENTS

Other instrument perturbations have been reported under LER's 1-90-006, 2-89-017, 1-87-017, and 2-86-020.

EIIS COMPONENT IDENTIFICATION

System/Component EIIS Code

Primary Containment Isolation System JM
High Pressure Coolant Injection BJ
Reactor Protection System JE
Emergency Diesel Generator EK
Reactor Core Isolation Cooling System BN
Residual Heat Removal/Low Pressure Coolant Injection BO
Core Spray BM
Standby Gas Treatment System BH
Safety Relief Valve */RV

* No EIIS System Identifier Found

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Figure "Attachment A" omitted.

ATTACHMENT 1 TO 9108220114 PAGE 1 OF 1

Form 244 CP&L
Carolina Power & Light Company

Brunswick Nuclear Project
Company Correspondence P. O. Box 10429
Southport, N.C. 28461-0429

August 16, 1991

FILE: B09-13510C 10CFR50.73

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555

BRUNSWICK STEAM ELECTRIC PLANT UNIT 1
DOCKET NO. 50-325
LICENSE NO. DRP-71
LICENSEE EVENT REPORT 1-91-018

Gentlemen:

In accordance with Title 10 of the Code of Federal Regulations, the enclosed Licensee Event Report is submitted. This report fulfills the requirement for a written report within thirty (30) days of a reportable occurrence and is submitted in accordance with the format set forth in NUREG-1022, September 1983.

Very truly yours,

J. W. Spencer, General Manager
Brunswick Nuclear Project
GT/

Enclosure

cc: Mr. S. D. Ebneter
Mr. N. B. Le
BSEP NRC Resident Office

*** END OF DOCUMENT ***